A Neutronic Feasibility Study for LEU Conversion of the WWR–M reactor at Gatchina [*)]

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ABSTRACT

In this report we present the results of computations of the full scale reactor core with HEU (90%), MEU (36%) and LEU (19.75%) fuel. The reactor computer model for the MCU RFFI Monte Carlo code includes all peculiarities of the core. Calculations show that a uranium density of $3.3 gU/cm^3$ of MEU (36%) fuel and $8.25 gU/cm^3$ of LEU (19.75%) in WWR–M5 fuel assembly (FA) geometry is required to match the fuel cycle length of the HEU (90%) case with the same end of cycle (EOEC) excess reactivity. For the equilibrium fuel cycle the fuel burnup and poisoning, the fast and thermal neutron fluxes, the reactivity worth of control rods were calculated for the reference case with HEU (90%) FA and for the MEU and LEU FA. The relative accuracy of this neutronic feasibility study of fuel enrichment reduction of the WWR–M reactor in Gatchina is sufficient to start the fabrication feasibility study of MEU (36%) WWR–M5 fuel assemblies. At the present stage of technology it seems hardly possible to manufacture LEU (19.75%) fuel elements in WWR–M5 geometry due to too high uranium density. Only a future R&D can solve the problem.

1. Introduction

1.1. In this work an attempt to study the possibility of conversion of the Gatchina research WWR–M reactor to low enriched fuel (19.75%) is made. In the previous report[1] we gave a brief description of the Gatchina WWR–M reactor, as well as of the assemblies with WWR–M2 (36% enriched) and WWR–M5 (90%) fuel elements, on which the critical experiments were carried out[2]. Detailed heterogeneous calculations of the benchmark reactivity using the Monte Carlo code MCU RFFI[3-5] were performed. A significant negative contribution to the reactivity of the heterogeneous effects was found. The accuracy of reactivity calculations was better than 0.3% for a wide range of the $H/^{235}U$ ratio and enrichment. Thus, we can recommend the code MCU RFFI for HEU–LEU neutronic computation. In the Ref.[1] the calculations were performed also for the full scale reactor. The contribution to the reactivity of the main reactor structure elements were investigated and Be poisoning was also estimated.

1.2. Brief description of the WWR-M reactor. In order to study the feasibility of the enrichment reduction in the WWR–M reactor a typical configuration of the core with a source of ultra cold neutrons (UCNS) is chosen (Fig.1)[1]. The core loading includes 144WWR–M5 fuel assemblies (FA) with (UO_2+Al) fuel element (FE). The FE thickness is equal to 1.25*mm* and the thickness of the meat is 0.39*mm* (Fig.2). The dimensions of the FE in the FA are presented in Tab.1. The high density of ^{235}U in WWR-M5 fuel elements ($66g^{235}U$/FA, 90% of enrichment) allows one to put experimental devices of significant size inside the core and to carry out unique fundamental investigations with cold, ultra cold and polarized neutrons[6].

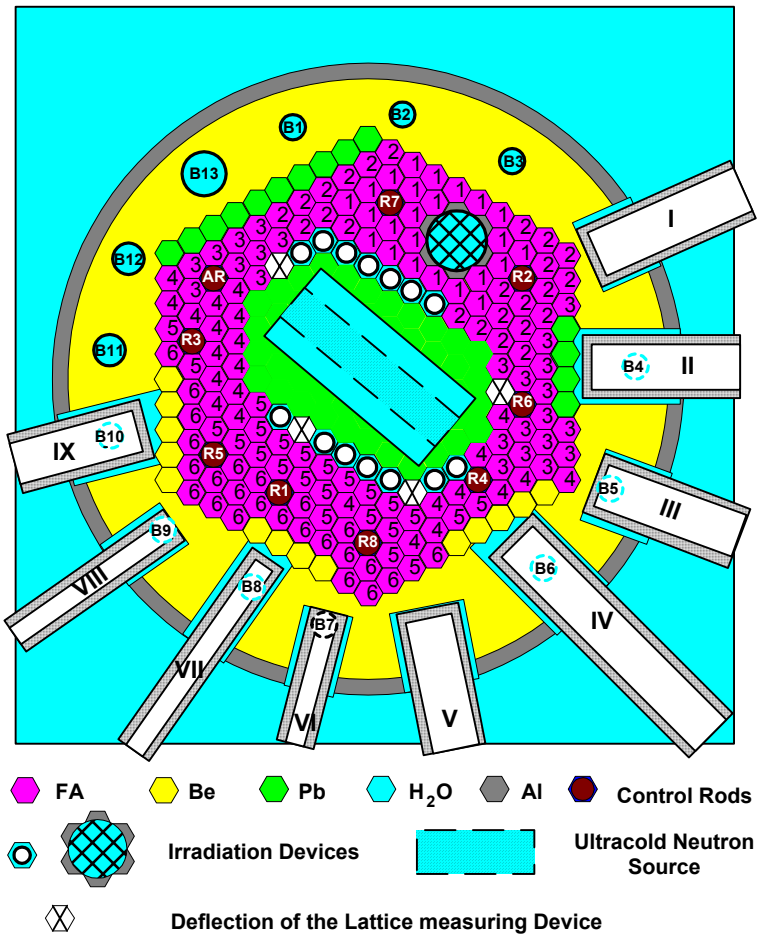

Fig.1.WWR-M core for LEU-HEU conversion study.

1-6 are steps of fuel burnup . R1-R4,R7,R8 are regulating rods. R5,R6 are safety rods. AR is automatic regulating rod.

A source of cold and ultra cold neutrons (UCNS) is situated in the middle of the core. It occupies 69 cells. The volume of the zirconium chamber with liquid hydrogen is about $1l$. It is cooled by helium. For irradiation of materials (various kinds of steel) ampoules of one cell size are used. The loading of 14 ampoules corresponds to the usual volume of tests. The ampoules are placed near the minimum shield thickness of UCNS. The device for irradiation of materials by fast neutrons occupies 7 cells. In order to reduce the induced activity it has a cadmium screen. Fourteen beryllium blocks are situated in front of horizontal channels to reduce the flux of fast neutrons at the channel bottoms and thus to increase their resource. For reduction of gamma radiation, 4 blocks of lead are placed opposite to the horizontal channel II. Opposite to the graphite thermal column a row of lead blocks is placed to reduce its gamma heating.

Table 1: WWR–M5 FA specification ($H_{FA} = 50cm$, $S_C = 10.609cm^2$, $V_{FA} = 530.44cm^3$, lattice spacing $a = 3.5cm$)

1	FE wall/clad/meat thickness, *mm* FE dimension hex/cyl ··· /rod, *cm*	1.25/0.43/0.39 3.35/2.79/2.23/1.67/1.11/0.55 [1)]
2	Meat ratio in cell, $\omega_M = S_M / S_C$ Calculated H_2O ratio, $\omega_{H_2O}^{TH}$	0.12716 0.5758

[1)] Central rod diameter was chosen equal to 0.6075*cm*, using the condition: $\omega_{H_2O}^{TH} = \omega_{H_2O}^{EXP}$.
The diameter of meat in a central WWR–M5 rod is 1.55*mm*.

The beryllium reflector is cooled by water, flowing in vertical channels. The water ratio in beryllium is 2.5%. In the reflector 9 horizontal channels and 13 vertical ones are available.

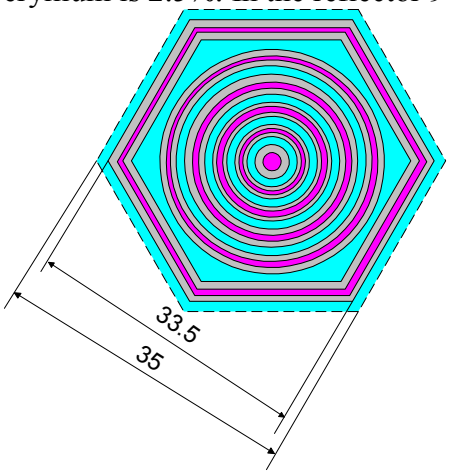

Fig.2. WWR - M5 Fuel Assembly
(the dimensions of FE are presented in Tab.1)

The horizontal channels II, IV, VII and IX reach the core, and the horizontal channels I, III, V, VI and VIII penetrate into the reflector by 10cm . Holes for the vertical channels B1,2,3,11,12 and 13 thread the whole reflector, and B4,5,6,7,8,9 and 10 reach only the horizontal channels. Channel B7 is empty, the other vertical channels are filled with water.

The control system of the reactor consists of 9 control rods containing B_4C. The places at which the neutron flux will be compared after enrichment reduction are the 1*l* volume of UCNS, the total volume of ampoules, the 7 cell water trap for irradiation, typical vertical channels B2, B8, B13 and the bottoms of typical horizontal channels II, IV, VIII (Fig.1).

1.3. Equiblirium fuel cycle. The fuel burnup model was an equilibrium cycle in which 24 fresh FA were loaded at the beginning of each cycle and discharged after remaining in the core for 6 cycles. The fuel management strategy used in this analysis was to load the lowest burnup fuel into the region of the core with the lowest flux and gradually move the fuel into the highest flux. This in/out strategy[7] provides a flatter energy release distribution. The reactor is running for 21 full power days (*fpd*) at a power level of 18MW and then is shut down for 14 days. The fuel discharge burnup at the end of the equilibrium cycle (EOEC) is 29%. The positions of 6 groups of FA with different burnup are shown in Fig.1.

At the cycle length t_c=*21 fpd* and power P=18MW the total energy produced per cycle is Pt_c=*378MWd/cycle* . The 24 discharged FA have a mean energy produced per FA equal to *15.8MWd/FA*. This value is close to the quantity *16.7MWd/FA* averaged over the last 20 years for the Gatchina reactor WWR–M[8]. For 6 *cycle/y* the total number of FA used per year is *144FA/y*. The MEU(36%) and LEU(19.75%) cores must use the same number of FA per year to operate.

In the calculations we have neglected the axial dependence of burnup. This axial distribution deviates from unity about 20% . With average 15% burnup the axial deviation in fuel density will be about ±3% . We can suggest that such a small deviation will lead to a small difference in

reactivity. Direct calculations give the negligible difference: –0.16(6)%. For MEU(36 %) and LEU(19.75 %) fuel this effect is even less due to lower burnup.

2. Results of fuel conversion study

2.1. HEU(90%) fuel. Let us calculate the reactivity of a fresh core with HEU fuel and losses due to burnup and reactor poisoning. We have used the burnup model with one stable fission product. The HEU fuel densities in meat and per WWR-M5 FA are shown in Tab.2. Values of K^{het}, reactivity and absorption in ^{235}U, ^{238}U for a fresh core were computed with MCU RFFI. They are also shown in Tab.2. The influence of ^{239}Pu is also included.

The calculation of Sm poisoning was performed for ^{149}Sm and ^{151}Sm having high absorption cross section and high fission yield. The parents of Sm isotopes are ^{149}Pm ($T_{1/2}$=*2.212(2)d*) and ^{151}Pm ($T_{1/2}$=*1.183d*). After the reactor is shut down, promethium continues to decay but samarium produced does not burn up and the Sm concentration is increased. This is the reason why the loss of reactivity due to Sm at the beginning of equilibrium cycle (BOEC) is nearly two times higher than at EOEC (Tab.3).

Table 2: Comparison of HEU(90%), MEU(36%) and LEU(19.75%) fuel

1	Fuel assembly, FA	WWR–M5H	WWR–M5M	WWR–M5L
2	Enrichment	HEU(90%)	MEU(36%)	LEU(19.75)
3	U Meat density, g/cm^3	1.087	3.30	8.25
4	^{235}U meat density, g/cm^3	0.9785	1.188	1.629
5	^{235}U(WWR–M5 …)/^{235}U(WWR–M5H)	1	1.214	1.665
6	^{235}U, g/FA	66	80.1	109.9
7	^{238}U, g/FA	6.6	141.5	445.4
	Fresh core			
8	Reactivity, % [K_{eff}]	12.91(4) [1.1483(5)]	11.73(4) [1.1329(5)]	10.50(5) [1.1170(6)]
9	Absorption in ^{235}U/n_f	0.568	0.559	0.552
10	Absorption in ^{238}U/n_f, 10^{-2}	0.452	3.84	6.99
11	$<\Sigma_a(^{238}U)>/<\Sigma_a(^{235}U)>$, 10^{-2}	0.796	6.87	12,7
	Discharged fuel			
12	$N(^{239}Pu)/N_0(^{239}Pu)$, %	0.18	1.35	1.91
13	Burnup, y_6, %	29.0	23.4	16.81
14	Average burnup, $\bar{y}$ (EOEC), %	14.9	11.85	8.50

By taking into account ^{135}Xe (and ^{105}Rh) poisoning the thermal neutron flux Φ_i in the group is changing. One needs 2–3 iterations to stabilize the neutron flux and Xe concentration. At BOEC xenon is absent. The reactivity loss due to (^{135}Xe +^{105}Rh) poisoning is sufficiently high: $\Delta\rho_{Xe}(EOEC) = -3.85(9)\%$. The final core reactivity excess is *+3.0(1)%*. The additional beryllium poisoning owing to ^{6}Li and ^{3}He formation is near to 2% (see for more details[8]). Change of reactivity with time is shown in Fig.3.

2.2. MEU (36%) fuel. For MEU (36%) fuel we choose the meat with $(UO_2 + Al)$ ceramic in Al matrix. The geometry of WWR–M5M FA with MEU fuel is chosen exactly the same as the WWR–M5 FA geometry with HEU (90%) fuel. Only the uranium density in the meat must be changed. After several preliminary calculations we find that the density of $3.3 gU/cm^3$ matches the fuel cycle length of the HEU (90%) case for the same EOEC excess reactivity. The density of UO_2 is $10.96 g/cm^3$ [9] and for the density of U in UO_2 we have $9.66 g/cm^3$. The density of $3.3 gU/cm^3$ is equivalent to 34 volume % of UO_2 in the Al matrix. This value seems acceptable. Nine thin–walled FA with density $3 gU/cm^3$ in $(UO_2 + Al)$ meat were fabricated and were burned up to 57% (at 21% enrichment)[10]. For the enrichment 36% it is equivalent to 33% burnup. This value is above the 24% burnup needed for WWR–M5M FA. Therefore the feasibility to fabricate the WWR–M5M FA with density of uranium $3.5 gU/cm^3$ seams realistic. The final conclusion could be reached after proper R & D work.

A comparison of fresh cores with HEU (90%) and MEU (36%) fuel is shown in Tab.2. At 36% enrichment the fuel would have 0.4% ^{234}U and 63.6% ^{238}U. The content of ^{235}U in WWR–M5M is increased only by a factor of 1.21, but the content of ^{238}U is 21.4 times higher. Due to the higher capture in ^{238}U the reactivity of a fresh core with WWR–M5M FA is 1.2% lower than for HEU (90%) FA. The total absorption in ^{238}U is increasing not by 21

Table 3: Fuel burnup and poisoning .

HEU (90%)					
1	Fresh core, ρ_0^{het}, %	12.91(6)			
		BOEC ($\bar{y}_F = 10\%$)		EOEC ($\bar{y}_F = 15\%$)	
		ρ_i^{het}, %	$\Delta\rho_i^{het}$, % [1)]	ρ_i^{het}, %	$\Delta\rho_i^{het}$, %
2	Fuel burnup	9.33(6)	–3.58(9)	7.60(6)	–5.31(9)
3	^{239} Pu contribution	9.34(6)	+0.01(9)	7.71(6)	+0.11(9)
4	Sm poisoning	7.70(4)	–1.64(7)	6.86(6)	–0.85(9)
5	(^{135}Xe +^{105}Rh) poisoning	—	—	3.01(6)	–3.85(9)
MEU(36%)					
6	Fresh core, ρ_0^{het}, %	11.73(4)			
		BOEC ($\bar{y}_F = 7.9\%$)		EOEC ($\bar{y}_F = 11.9\%$)	
		ρ_i^{het}, %	$\Delta\rho_i^{het}$, % [1)]	ρ_i^{het}, %	$\Delta\rho_i^{het}$, %
7	Fuel burnup	8.95(6)	–2.78(7)	7.55(6)	–4.18(7)
8	^{239}Pu contribution	8.90(6)	–0.05(9)	7.71(6)	+0.16(9)
9	Sm poisoning	7.43(4)	–1.47(7)	6.63(6)	–1.08(9)
10	(^{135}Xe +^{105}Rh) poisoning	—	—	2.97(6)	–3.66(9)
LEU(19.75%)					
11	Fresh core, ρ_0^{het}, %	10.47(5)			
		BOEC ($\bar{y}_F = 5.7\%$)		EOEC ($\bar{y}_F = 8.5\%$)	
		ρ_i^{het}, %	$\Delta\rho_i^{het}$, % [1)]	ρ_i^{het}, %	$\Delta\rho_i^{het}$, %
12	Fuel burnup	8.50(4)	–1.97(7)	7.62(5)	–2.85(7)
13	^{239}Pu contribution	8.62(4)	+0.12(6)	7.88(5)	+0.26(7)
14	^{240}Pu contribution	8.40(4)	–0.22(6)	7.56(5)	–0.32(7)
15	Sm poisoning	7.31(4)	–1.09(6)	6.56(5)	–1.00(7)
16	(^{135}Xe +^{105}Rh) poisoning	—	—	2.99(5)	–3.57(7)

[1)] $\Delta\rho_i = \rho_i^{het} - \rho_{i-1}^{het}$.

times but only by 8.5 times because the resonance absorption is partly blocked (see row 10 of Tab.2). The ratio of absorption $\delta_{Pu} =< \Sigma_a(^{238}U)\Phi / \Sigma_a(^{235}U)\Phi >$ responsible for ^{239}Pu production is also 8.6 times higher.

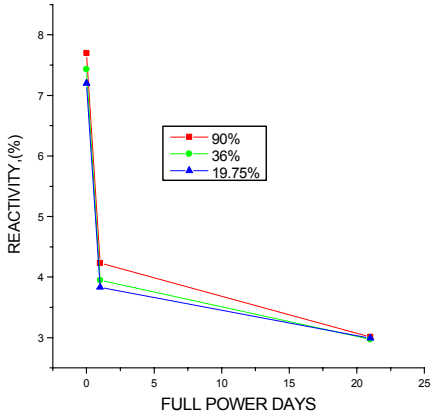

Fig.3 WWR-M Reactivity Rundown Curves

The energy produced by the fission of ^{239}Pu and ^{238}U is 2.4% of the energy produced by fission of ^{235}U (9MWd). By production of 369MWd of energy the ^{235}U total burnup is only $y_6^F = 23.4\%$ due to the higher incident concentration of ^{235}U in the meat of WWR–M5M FA. Thus, to match the fuel cycle length of HEU (90%) fuel we need a lower burnup. The reactivity losses due to burnup and poisoning for the core with WWR–M5M FA are shown in Tab.3. Comparing the HEU (90%) case the loss of reactivity by MEU fuel burnup is less: 4.2(1)% at EOEC instead of 5.3(1)% for HEU. The ^{239}Pu contribution to reactivity is still small. The *Sm* poisoning is nearly the same. The *Xe* poisoning is also the same. Fig.3 demonstrate the change of reactivity versus the number of *fpd*. At the EOEC the remaining excess of reactivity is the same for both cases.

The density of $3.3 gU/cm^3$ in meat of MEU (36%) fuel is enough to match the HEU (90%) core fuel cycle. Maybe easier is to use *U–Mo (9w.%)* fuel in an *Al* matrix instead of UO_2. The calculations show that in this case we need uranium density of $3.546 gU/cm^3$ in meat of MEU (36%) fuel to match the HEU (90%) core fuel cycle. The multiplication factor at EOEC is $K(EOEC) = 1.0296(13)$ and the reactivity $\rho(EOEC) = 2.87(12)\%$ is inside the error bars of the case of UO_2 MEU fuel (see Tab.3).

2.3 LEU (19.75%) fuel. In this case a density is required which is higher than can be achieved in ($UO_2 + Al$) ceramic. In 1996 the extremely interesting program of creation of high density fuel with $\gamma_U = 8-9 gU/cm^3$ in the meat was started in the USA[11]. According to this program the plates with an alloy *U–Mo* (9 weight %) in an *Al* matrix were fabricated for testing in the reactor[12]. The density of the alloy is $17.0 g/cm^3$, the weight fraction of uranium in the dispersant is $W_U^D = 0.91$, thus the density of uranium in alloy is $15.5 gU/cm^3$. (The first Russian atomic power station ($5MW(el)$) started at 1954 used *U–Mo* (9*w*.%) metallic grist in the matrix of fuel elements[13]).

After several preliminary calculations we find that the density of $8.25 gU/cm^3$ (or $1.629 g^{235}U/cm^3$ in fuel meat) in the WWR–M5 geometry FA, match the fuel cycle length of HEU (90%) for the same EOEC excess reactivity. It is equivalent to 53 volume % of *U–Mo* (9%) fuel in the *Al* matrix. The plates for in–pile irradiation having such a high volume % of fuel in the meat were fabricated at ANL using powder technology. Such a value could be easily achieved by Western rolling plates technology. For the Russian extrusion of tubes technology[14] no similar data where published. Serious R&D work is needed to show the

feasibility of high density fuel production by extrusion technology. In–pile irradiation tests are also needed. Here we present the results of neutronic calculations only.

A comparison of fresh cores with LEU (19.75%), MEU (36%) and HEU (90%) fuel is shown in Tab.2. At 19.75% enrichment the fuel would have 0.22% ^{234}U and 80.0% ^{238}U. Compared with HEU (90%) the content of ^{235}U in FA with $8.25 gU/cm^3$ of LEU (19.75%) fuel in meat (WWR–M5L) is 1.625 times higher. Compared with MEU (36%) the density of ^{238}U in the meat of WWR–M5L is 3.1 times higher. But absorption in the ^{238}U for a fresh core is only 1.8 times higher (row 11 in Tab.2). This is due to a strong block–effect in the resonances of ^{238}U. The neutrons are absorbed only in the thin layer of meat and the resonance absorption is approximately proportional to the value $\sqrt{N_M^8}$. The ratio $\delta_{Pu} = <\Sigma_a(^{238}U)\Phi/\Sigma_a(^{235}U)\Phi>$ responsible for the ^{239}Pu production is also 1.8 times higher than for MEU (36%) fuel.

The energy produced by the fission of ^{239}Pu and ^{238}U is 3.4% of the total energy produced by the fission of ^{235}U ($13 MWd$). By production of $W_5 = 365 MWd$ of energy, the total ^{235}U burnup in the $6th$ group of FA is only $y_6^F = 16.8\%$ according to the equation

$$y_6^F(LEU) = \left(y_6^F \gamma_5 / W_5\right)_{MEU} / \left(\gamma_5 / W_5\right)_{LEU} \qquad \textbf{(1)}$$

due to the higher incident concentration of ^{235}U in the meat of WWR–M5L. The reactivity losses due to burnup and poisoning for the core with WWR–M5L FA are shown in Tab.3. Compared with the HEU (90%) case, the loss of reactivity by LEU fuel burnup at EOEC is considerably less. The ^{239}Pu contribution to reactivity is still small. The *Sm* and *Xe* poisoning are nearly the same. Fig.3 demonstrates the change of reactivity versus the number of *fpd*. The uranium density of $8.25 gU/cm^3$ in the meat of LEU (19.75%) fuel is theoretically enough to match the HEU (90%) core fuel cycle.

3. Rod reactivity worth

All 9 control rods (Fig.1) are identical and could be placed in every cell of the core. One of them is an automatic regulating rod (AR); the control rods number 5 and 6 are safety rods (SR) and are always in the upper position. The other six control rods are regulating rods (RR). Efficiency of control rods and interference effects are shown in Tab.4. First in Tab.4 are shown the sums of reactivity of *m* single rods. The reactivity worth of a single rod is a difference of reactivities for two situations: 1) all CR are withdrawn and 2) one of CR is inserted, all other are withdrawn. In the next row of Tab.4 is presented the reactivity worth for *m* CR inserted together. The ratio of this reactivity to the previous one demonstrates the interference of control rods. It increases with *m* up to 1.4 for HEU fuel. The worth of 6RR must be high enough to compensate the reactivity difference between $\rho^{het}(BOEC)$ without Xe and $\rho^{het}(EOEC)$ with Xe.

For MEU and LEU fuel the reactivity worth of 2SR is nearly by the same as for HEU (90%) core. For MEU fuel the reactivity worth of 6RR is 3% less. But the reactivity change $\rho^{het}(BOEC) - \rho_{Xe}^{het}(EOEC) = 4.46(7)\%$ (Tab.3), which must be compensated by 6RR, is less than in the case of HEU (4.7%). For LEU (19.75%) fuel the reactivity worth of 6RR is less by $0.61(11)\%$. But the reactivity change $\rho^{het}(BOEC) - \rho_{Xe}^{het}(EOEC) = 4.3(1)\%$ (Tab.3), which must be compensated by 6RR, is less than in the case of HEU (4.7(1)%). The compensation ability of the control rods system is not worse by HEU to LEU conversion.

Table 4: Worth of control rods $\Delta\rho^{het}{}_m(\%)$

		AR	SR	RR	All
	Number of rods, m	1	2	6	9
HEU(90%)	Sum of m single CR	0.57(6)	2.24(9)	5.73(15)	8.54(18)
	Reactivity of m CR	0.57(6)	2.40(7)	7.29(7)	11.87(7)
	Interference	1	1.07(5)	1.27(4)	1.39(3)
MEU(36%)	Sum of m single CR	0.78(9)	2.22(13)	5.78(22)	8.78(27)
	Reactivity of m CR	0.78(9)	2.39(9)	7.17(9)	11.53(9)
	Interference	1	1.08(7)	1.24(5)	1.31(4)
LEU(19.75%)	Sum of m single CR	0.64(8)	1.97(12)	5.17(20)	7.78(25)
	Reactivity of m CR	0.64(8)	2.34(8)	6.68(9)	10.60(9)
	Interference	1	1.19(8)	1.29(5)	1.36(5)

4. Neutron fluxes

Neutron fluxes calculated with MCU RFFI are shown in Tab.5. The computations were performed for thermal $(< 0.625eV)$ and fast $(> 0.821MeV)$ neutrons at 9 specified experiment positions for a fresh core with HEU (90%), MEU (36%) and LEU (19.75%) fuel and also at the EOEC. Fast and thermal neutron fluxes were calculated for all positions but for some of them only one flux is interesting. So, for the ampoules and irradiation device only the fast neutron flux is interesting and for the UCNS one needs thermal neutron flux.

At $18MW$ the thermal neutron flux for UCNS is rather high: about $3\cdot10^{14}n/cm^2s$. This flux is also high for vertical channel B8: above $2\cdot10^{14}n/cm^2s$. For the irradiation device the thermal neutron flux is suppressed by the *Cd* shield. In the inner part of the core the fast neutron flux for ampoules rises up to $6\cdot10^{13}n/cm^2s$ for HEU, MEU and LEU fuel. The neutron fluxes for EOEC are nearly the same as for a fresh core.

The change of the neutron fluxes by HEU (90%) to MEU (36%) conversion is insignificant. For example for EOEC the thermal neutron flux at UCNS is 4(1)% less for MEU core and for B8 this flux is 9(1)% less. The change of fast neutron flux for ampoules is negligible: 1.0(6)%. The change of the neutron fluxes by HEU (90%) to LEU (19.75%) conversion is also small. For EOEC the thermal neutron flux at UCNS is 7(2)% less for LEU core; for B8 this flux is 13(1)% less. The change of fast neutron flux in the ampoules is: 3.5(6)%. Thus, by HEU to LEU conversion the decrease of neutron fluxes is acceptable.

Table 5: Neutron fluxes at 18MW calculated with MCU RFFI code: fast (>0.821MeV) in 10^{13}n/cm^{2}s, thermal (<0.625eV) in 10^{14}n/cm^{2}s.

		Neutron flux	HEU (90%)		MEU (36%)		LEU (19.75%)	
			Fresh core	EOEC	Fresh core	EOEC	Fresh core	EOEC
1	B2	Fast Thermal	0.28(1) 0.74(1)	0.25(1) 0.64(1)	0.30(1) 0.71(1)	0.22(1) 0.62(1)	0.27(1) 0.66(1)	0.24(1) 0.58(1)
2	B8	Fast Thermal	1.42(4) 2.08(3)	1.94(4) 2.83(3)	1.50(5) 2.16(4)	1.82(6) 2.58(4)	1.45(4) 2.00(3)	1.80(4) 2.46(3)
3	B13	Fast Thermal	0.50(1) 0.93(1)	0.57(1) 1.05(1)	0.52(1) 0.92(1)	0.54(1) 0.99(1)	0.49(1) 0.87(1)	0.51(1) 0.92(1)
4	H2	Fast Thermal	3.45(8) 0.83(2)	4.18(9) 1.02(2)	3.4 (1) 0.74(2)	4.4 (1) 0.95(3)	3.55(9) 0.69(2)	3.99(9) 0.79(2)
5	H4	Fast Thermal	4.02(9) 1.49(2)	3.70(9) 1.47(2)	3.9 (1) 1.42(3)	3.8 (1) 1.44(3)	3.9 (1) 1.39(3)	3.80(9) 1.34(2)
6	H8	Fast Thermal	2.7 (1) 2.04(4)	2.04(9) 1.74(4)	2.7 (2) 1.85(5)	2.3 (1) 1.66(5)	2.7 (1) 1.64(4)	2.00 (9) 1.48(3)
7	Ampoules	Fast Thermal	6.28(2) 0.694(2)	6.34(2) 0.700(2)	6.31(3) 0.651(3)	6.28(3) 0.654(3)	5.82(3) 0.579(2)	6.12(2) 0.575(2)
8	Irrad. Device	Fast Thermal	2.83(4) 0.094(2)	3.89(5) 0.129(3)	2.86(6) 0.093(3)	3.69(7) 0.122(3)	2.90(5) 0.090(2)	3.68(5) 0.115(2)
9	UCNS	Fast Thermal	3.34(4) 2.92(2)	3.46(4) 2.96(2)	3.32(5) 2.84(3)	3.39(5) 2.84(3)	3.19(4) 2.74(2)	3.21(4) 2.74(2)

Conclusions

Calculations with MCU RFFI code show that the uranium density of 3.3gU/cm^3 of MEU (36%) fuel in WWR–M5 FA geometry is required to match the fuel cycle length of the HEU (90%) case with the same EOEC excess reactivity. The ($UO_2 + Al$) MEU fuel would have 34 volume % of UO_2 in the Al matrix. The burnup in the discharged MEU FA (WWR–M5M) is 23.4%. The change of neutron fluxes is variable but insignificant: less then 10%. In case of MEU (36%) fuel the reactivity worth of control rods is nearly the same as for HEU (90%) fuel. The possibility to use such a fuel must be confirmed by corresponding R&D work. Maybe the [U+Mo(9$w.\%$)] fuel with $3.5 gU/cm^3$ in meat (23 volume % in Al matrix) will be more easy for fabrication.

Due to high loading of the core of the Gatchina WWR–M reactor with experimental facilities its reactivity excess for fuel burnup is rather low. Therefore a high uranium density of $8.25 gU/cm^3$ in the meat of LEU (19.75%) fuel in WWR–M5 FA geometry matches the fuel cycle of HEU (90%) core with the same EOEC excess reactivity. The $(U + Mo(9\%))$ LEU fuel requires 53 volume % in Al matrix. If the thickness of cladding of FE could be diminished from $0.43 mm$ to $0.38 mm$, the volume % of LEU fuel could be 42%. This value is still to high for extrusion technology. Very serious R&D work must be performed to demonstrate the fabrication feasibility of such a WWR–M5L FA.

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